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Preliminary Calculations of Shutdown Dose Rate for the CTS Diagnostics System

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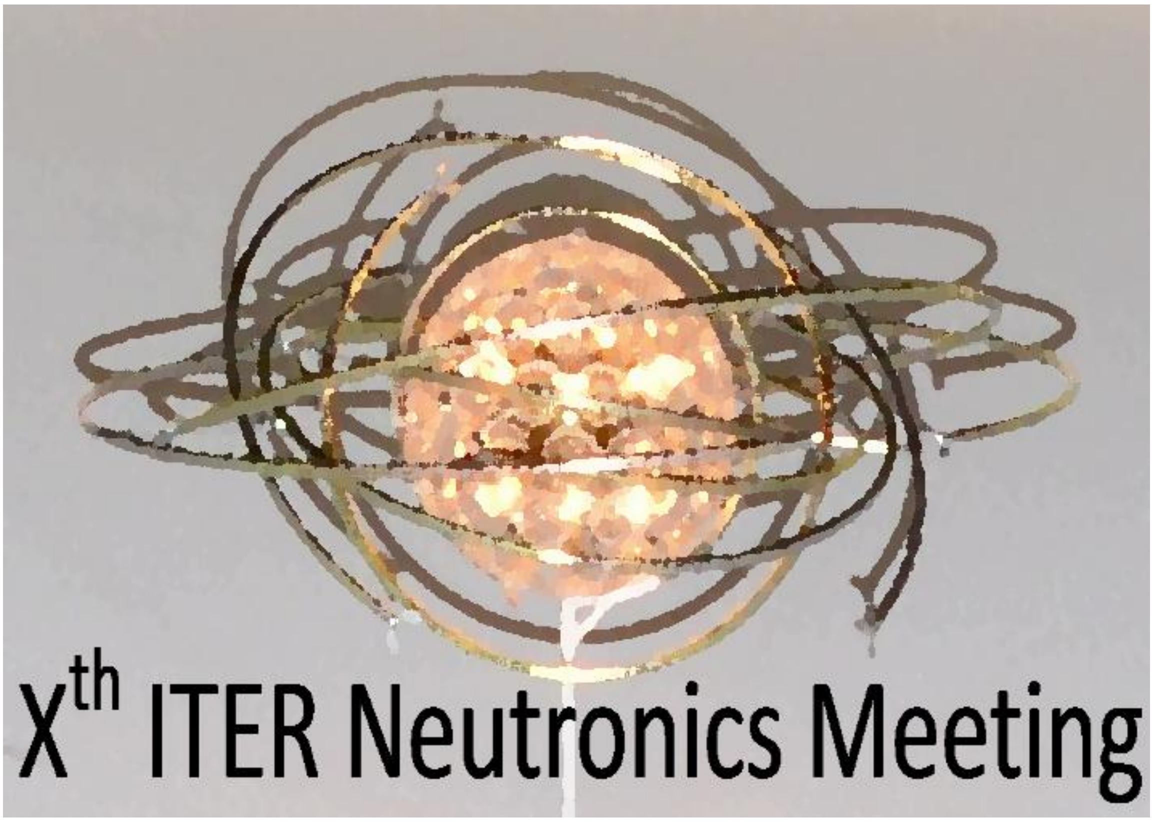

Book of Abstracts

ITER Status and Priorities for Neutronics

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A brief summary of the status of ITER construction and manufacturing will be presented with some short comments on the new organisation.

The priorities for ITER neutronics as expressed by the most recent ITER Council are the TF coil heating and the shut-down dose rates in the cryostat. A substantial amount of recent work has been conducted and in these areas which is reflected in the number agenda of this meeting. The future areas of work are described.

There are several other issues which are also important and the evolving role of nuclear analysis will be described. A significant new aspect of nuclear analysis is the experimental verification of the analyses and construction. This must be supported by benchmarks and verification of nuclear data.

To increase the contributions from Domestic Agencies it is important to explore the use of a wider range of nuclear analysis codes. The development of these codes and the other supporting software is briefly introduced.

Neutronics Activities at the University of Wisconsin in Support of ITER Design

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The neutronics group at the University of Wisconsin is carrying out several neutronics activities to support the ITER project. We will summarize these activities and discuss the technical findings. These activities are listed below:

- 3-D neutronics for blanket modules to generate detailed nuclear heating maps needed for CFD and structural analyses
- 3-D neutronics at several locations with excessive vacuum vessel heating to assess impact of proposed design changes.
- Assessment of sensitivity of ITER neutronics results to recently released photon data.
- Generating a full clean CAD model that is consistent with the latest MCNP version of C-lite to be used with CAD-based codes such as DAG-MCNP and ATTILA
- Preparation of a clean CAD model that includes detailed geometry up to the bio-shield and incorporates detailed port plugs to determine full shutdown dose map inside the bio-shield using R2S-ACT.
- Analysis with detailed heterogeneous BM and VV with FSH to assess needed correction factors for TFC heating.
- Determining best estimate for TFC total heating based on latest models and accounting for correction factors in outboard from detailed analyses.
- CAD preparation for the TCWS with detailed water path from the UP pipe chase into FW and SB detailed geometry and out of the tokamak.
- Performing water activation analysis to determine the accurate source term from ^{16}N and ^{17}N for L3/L4 shielding analysis.
- Performing DAG-MCNP calculation to compare to ATTILA results using the same CAD model with a diagnostic port plug.

Overview of Neutronic and Radiological Analyses at UNED

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TECF3IR group of UNED has recently produced relevant radiological and neutronic results in the framework of different contracts with ITER IO and F4E. This presentation summarizes the main results stressing the impact they may have on ITER systems.

The heating of poloidal and toroidal field coils due to N-16 gammas from activated cooling water has been computed. Opposite to a similar analysis performed last year, the new design contains divertor cooling pipes outside the cryostat, resulting in a much higher energy deposition into coils and their casings, with peak values of 1 kW/m^3. These values are still much lower than those by prompt neutrons, though.

The impact during cask transfer operations of the port cell lintel to doses in the port gallery has been assessed. Considering the transport of four activated blanket shield modules, the effect of lintel penetrations on doses in the gallery has been studied, and effective shielding designs proposed, achieving doses as low as 25 μSv/h at floor level of the gallery.

The dose field in the ICH port has been computed, for both reactor operation and also shutdown, considering the activation of both concrete structures and the ICH device as well. The results show a hot region between bioshield plug and the internal neutron shield, and beyond that point a flatter field resulting in integrated absorbed doses (mainly gamma) of around 1 kGy(Si) for SA-2 irradiation scenario. Shutdown dose rates for 1 day cooling, take values of hundreds of μSv/h up to the local shielding, reduced to less than 10 μSv/h beyond that point.

This overview will also present other relevant results and the computational developments in the group, which allow to accurately perform these analyses of both prompt and residual dose fields. These issues will be dealt in depth during this meeting in individual presentations.

Overview of Recent ITER Related Neutronics Research Activities in China

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Supported by the projects related to ITER and fusion reactors, neutronics technologies in China have been greatly improved in recent years, including the development of neutronics codes and nuclear data libraries, establishment of neutronics facilities and experiments, and nuclear analyses.

A series of neutronics codes, featured to deal with complicated nuclear systems, have been developed by INEST·FDS Team, such as the Super Monte Carlo Calculation Program for Nuclear and Radiation Process (SuperMC). SuperMC is designated to perform the comprehensive neutronics calculation, taking the radiation transport as the core and including the depletion, radiation source term/dose/biohazard, material activation and transmutation, etc. The main usability features are automatic modelling of geometry and physics, visualization and virtual simulation and cloud computing service. SuperMC version 2.2 has been newly implemented to accomplish the transport calculation of neutron and photon, with automatic modelling module SuperMC/MCAM, visualized analysis module SuperMC/RVIS, and hybrid evaluated nuclear data library SuperMC/HENDL.

The project of High Intensity D-T Fusion Neutron Generator (HINEG) in China has been launched, mainly aiming to perform experimental studies related to fusion nuclear technology such as fusion neutronics and material testing. Phase I of this facility is to build an accelerator-based D-T fusion neutron generator (HINEG-I) with a neutron yield of 10^{12} n/s, which will be finished at the end of 2015. Phase II of this facility is to build an upgraded generator (HINEG-II) with a neutron yield of 10^{14}n/s. The design and pre-research of HINEG-II are progressing. Furthermore, A long term plan and conceptual design of the Volumetric Fusion Neutron Source (VFNS) for component engineering testing is under progress.

The neutronics analysis requirements of these procurements, and the Chinese TBM system, have been investigated and identified, and preliminary neutronics analysis is under way. INEST·FDS Team has coordinated the related Chinese institutions, who are responsible for ITER procurements related to neutronics analysis.

KIT's Activities Related to ITER and Fusion Neutronics

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This presentation provides an overview of the activities of the Karlsruhe Institute of Technology (KIT) in the field of fusion neutronics related to ITER. The activities include development works on computational tools and methods which are relevant to ITER neutronics, modelling works for ITER components, as well as design and performance related analyses on specific issues conducted in the frame of running ITER and F4E contracts. Complementary development works of interest to ITER, mainly performed in the frame of the EUROfusion project and specific F4E grants, are also briefly addressed. These include, among others, the development of a common European R2S tool, developed jointly with CCFE (UK) and UNED (Spain), experimental investigations on neutron detector systems for use in the TBM in ITER, and the development of specific nuclear data evaluations/libraries for radiation damage and gas production calculations.

Overview of Recent Progress in Neutronics Activities on JET in View of DT Operations in Support of ITER

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In view if the new DT campaign at JET (DTE2) planned in 2017- 2018, a project was launched to investigate technological areas relevant to ITER (EUROfusion WPJET3 – DT Technology). In this frame, several neutronics activities were planned in support of ITER, and have started already in 2014, including:

i) *14-MeV calibration of JET neutron detectors*. A 14 MeV neutron generator has been purchased and will be deployed, as the neutron source, inside the vacuum vessel by the Remote Handling system. The generator will be fully calibrated and characterized in a neutron laboratory using a number of different neutron detectors. A characterization/calibration strategy has been defined, supported by extensive neutron calculations, in order to achieve a target accuracy of ±10%. The whole calibration procedure is meant as a benchmark of the neutron calibration in ITER.

ii) *Neutronics experiments.* Both a Neutron streaming experiment (NSE) and a Shutdown dose rate experiments (SDR) are planned in the next DD/DT campaigns. For the NSE, additional experimental positions outside the bio-shield will be investigated. Thermoluminescent dosimeters used so far will be complemented with activation foils inside the torus hall for neutron fluence measurement. For the SDR experiment, a new position has been selected at the horizontal port where the ITER-like ICRH antenna is located, to investigate the SDR due to the neutron streaming through the antenna structures. All measurements will be used to validate neutronics codes and nuclear data used in ITER.

iii) Activation of ITER materials. Two in-vessel long term irradiation stations (LTIS) have been installed inside the vacuum vessel where real ITER in-vessel materials and diagnostics functional materials will be irradiated for post-irradiation analysis. About 150 dosimetry foils have been installed in the LTISs to be irradiated in the next DD campaign and then analysed to fully characterize the neutron spectrum (from 14 MeV to thermal) at LTIS positons.

Recent achievements and their relevance for ITER will be presented at the meeting and discussed.

See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia

Shutdown Dose Rate Benchmark Experiment at JET After DD Operations

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A shutdown dose rate benchmark experiment has been conducted at JET at the end of 2012 DD campaign[1] to validate shutdown dose rate codes used in ITER predictions. Dose rates have been measured outside the horizontal port of Octant 1 and along the mid-port from ~1 day to 6 months after DD shutdown. Five active detectors have been used and dominant isotopes have been identified as well. Cross –calibration studies have also been done. 3-D calculations with the recent versions of Advanced D1S (ENEA), R2Smesh (KIT), MCR2S (CCFE) and R2SUNED (UNED) have been performed using an updated Octant 1 MCNP model of JET based on CAD files. The results of the calculations have been compared with experimental data. Except some positions, a good agreement within the overall uncertainties has been found between calculations and measurements and among the codes. The results of the benchmark experiments confirm that the available calculation methodologies for shut-down dose rate assessments are capable of reproducing the measured quantities in the examined temporal range in a tokamak configuration within the experimental uncertainty margin provided that proper geometry and nuclear data are taken in to account. The results benchmark provide further confidence in the prediction capabilities of the R2S and D1S computational tools for application to ITER and future power reactors as well as useful outcomes for reducing uncertainties and preparing the future shutdown dose rate experiment during DTE-2 (EUROfusion WPJET3).

[1] This activity was carried-out within EFDA task JET JW12-FT-5.43

Updating of the In-vessel Components for the C-lite Model

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The ‘C-lite’ model is the current reference MCNP model of ITER for the neutronics community, and is used extensively for the nuclear analysis of ITER systems. As such, accurate and faithful representations of geometry in C-lite is paramount, in order to produce reliable results for key nuclear responses. The geometry associated with the in-vessel components is known to be particularly important for the accurate simulation of nuclear heating in the vacuum vessel and superconducting coils, and the shutdown dose rate in the ports. So called ‘semi-detailed’ in-vessel models of the in-vessel components are under development to replace the previous simplified representations and increase modelling fidelity. Prior to this work, the only semi-detailed MCNP models of in-vessel components available were in blanket rows 1-6 and 13-18 (which were developed at an earlier stage in the design process).

In the frame of an F4E Task Order, semi-detailed representations of the latest designs for blanket rows 7 – 12 were jointly developed by CCFE, KIT-ENEA, and Amec FW. Models were developed representing the first wall and shield block, cooling pipe manifolds, in-vessel coils and vacuum vessel interfaces. Where simplifications were necessary, density corrections were made to preserve mass and hence the neutronics behaviour of the system. Emphasis was placed on accurate geometry modelling for the inter-modular gaps, coil pockets and manifold recesses, which will strongly affect the transport of high energy neutrons between the modules. Since the models were to be used in Protection Important Activities and in the computation of responses for Protection Important Components (e.g. the vacuum vessel), thorough verification activities were undertaken and documented. These models were subsequently integrated into C-lite and used to obtain updated nuclear heating maps of the vacuum vessel. This presentation will detail the modelling and verification efforts, the outcomes of the task and lessons learnt.

This work was funded by F4E under contracts OMF-0466-01-04, OMF-0466-02-03, OMF-0466-03-05.

Nuclear Analysis for Equatorial Port 10

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In order to control, evaluate and optimise the performance of the ITER plasma a number of diagnostic systems are necessary. Equatorial port 10 (EP10) houses four of these diagnostic systems including the Poloidal Polarimeter (POPOLA), Plasma Position Reflectometer (PPR), Core Plasma Thomson Scattering (CPTS) and the Edge Thomson Scattering (ETS) systems. All apart from the PPR are in the pre-conceptual design review stage. In order to specify sufficient neutron and photon shielding, and to progress the design, nuclear analysis has been performed.

Using MCNP and CCFE's implementation of the Rigorous Two Step Method, MCR2S, the neutron flux, nuclear responses and shutdown dose rate (SDDR) in the EP10 port interspace have been calculated. The SDDR in the port interspace has been calculated for a number of shielding options, including a baseline and an optimised shielding design.

The optimised shielding design was shown to reduce the SDDR in the EP10 port interspace by approximately two orders of magnitude compared to the baseline design. However, as part of the ALARA argument it was also necessary to calculate the SDDR contributions from the activation of the neighbouring ports, port interspace diagnostic components and each of the three diagnostic drawers. This showed that the dominant contribution to the SDDR in the port interspace is due to the activation of the port interspace components, with still significant but lower contributions from the POPOLA drawer and lower port.

To further reduce the SDDR in the in the port interspace several proposals were made including: removal of mass from the port interspace where possible or make port interspace components out of low activation material, a redesign of the POPOLA optical paths to allow for more shielding to be placed in the port plug and additional shielding of the lower port to reduce neutron and photon cross-talk.

This work was funded by F4E under contract OMF-0466-01-02.

Nuclear Analysis of Diagnostics Equatorial Port #11 Integration

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ITER equatorial port #11 contains six large diagnostics apertures with the corresponding systems, hosted in three diagnostics drawers inside the port plug. This structure, named “equatorial port plug” is inserted in the equatorial vacuum vessel port extension. After the closure plate, the diagnostics systems continue along the port interspace, presenting shielding, valves, and optical, systems. The port interspace structure mounted on a trolley withstands these interspace components, and moves over rails. The totality of these components is removed through the aperture left after removing the bioshield plug.

With the scope of shutdown dose rates for port integration activities, a nuclear analysis has been performed considering all the above mentioned components inserted into Clite. SpaceClaim, MCAM, MCNP, ACAB and R2S-UNED codes have been used, with the cross-sections libraries FENDL2.1 and EAF2007 to determine the shutdown dose rates at port interspace after 10^6 s of cooling time.

A decoupling exercise have been carried out, blocking different each diagnostic aperture for neutron flux. In addition, for each neutron flux contribution, independent activation of reactor systems has been computed. In this way, a very valuable table has been obtained, identifying the dose contribution of each system activated by neutrons from each diagnostic apertures. It was identified that DSMs lack of shielding and NPA in-port neutron cross-talk were major responsible for that situation. Filling the DSMs empty volume with B4C allowed to reducing dose rates by a 70%.

Nuclear Shielding Challenges for Hands-on Maintenance of ITER Port Systems

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Nuclear shielding of the ITER tokamak encompasses several systems and interfaces in a complex radiation environment. Therefore any shielding design has to necessarily involve a series of structures, systems and components in a particular integrated approach. This is evident for the complex ex-vessel radiation environment with streaming and leakage of plasma neutrons and subsequent activation of ex-vessel structures which give raise to excessive shutdown dose rates (SDR) in areas, where hands-on maintenance of ITER port systems is foreseen.

Radiation streaming through gaps and penetrations is a key problem in any efficient shield design. The ITER design for the primary shield configuration of blanket and vacuum vessel system is far from optimal for its shielding function, due to the need of ongoing integration of several in-vessel components and port based equipment. The ITER port systems have to contribute to the shielding towards their respective port interspaces. The impact on the evolving radiation environment due to several recently studied design options at the periphery of port systems, as well as their modeling for nuclear analysis, will be presented. Implications regarding design integration and compliance with integrated shielding requirements and ALARA dose rate will be given.

Neutronic Calculations in Support of the ITER Radial Neutron Camera Design

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The ITER Radial Neutron Camera (RNC) is a fundamental diagnostic system of the ITER experiment: it's designed to detect the uncollided 14 MeV and 2.5 MeV neutrons, respectively from deuterium-tritium and deuterium-deuterium reactions, through an array of detectors located along collimated lines of sight. The RNC will provide time and spatially resolved measurements of several plasma parameters needed for fusion power estimation, plasma control and plasma physics studies. The line-integrated neutron fluxes are used to evaluate the local profile of the neutron emission and therefore the total neutron yield, fusion power, neutrons and alpha source profile; moreover, the ion temperature profile and the fuel ratio in plasma core can be derived from neutron spectral measurements.

A 3D MCNP model of the RNC present layout has been developed. It's composed by two fan-shaped collimating structures viewing the plasma radially through cut-outs in the diagnostic shielding modules of the ITER Equatorial Port 1: the in-port system, consisting of two sets of four channels located inside the port for upper and lower plasma edge coverage, and the ex-port system consisting of a massive shielding block placed in the Port Interspace, hosting two sets of 10 collimators each, for plasma core measurements. Successively, the RNC model has been integrated in the latest release of the ITER C-lite model, along with other diagnostic systems that are foreseen in the Equatorial Port 1. Preliminary calculations of the neutron and gamma fluxes and spectra at the in-port detectors positions and studies aimed at the optimization of the collimators layout, are presented[2].

[2]The work leading to this publication has been funded partially by Fusion for Energy under the Specific Grant Agreement F4E-FPA-327 SG03.This publication reflects the views only of the author, and Fusion for Energy cannot be held responsible for any use which may be made of the information contained therein. This work was also supported by the grant Pest-OE/SADG/LA0010/2013 from "Fundação para a Ciência e a Tecnologia".

Neutronics and Radiation Analyis for XRCS Survey Sight Tube

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Neutronics and radiation analysis for XRCS survey sight tube has been carried out in order to estimate the neutronic parameters and personal dose rate contributed by the said system. The radiation shield has been designed and optimized to reduce the flux and dose to a level which meet ALARA principle. Activation and rad-waste analysis are also performed in detail for different kind of sight tube materials, which enable the designers to select proper material for the sight tube. The details of the analysis and the results will be presented in the meeting.

Shielding Analysis for Port 18 with WCCB-TBM Set

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We conducted the shielding calculations for the equatorial port #18 by using MCNP, FENDL-2.1 and the C-lite model. We inserted the identical two Japanese WCCB-TBM sets in the equatorial port instead of Korean HCCR-TBM set to the C-lite model. We created the simple CAD data on the WCCB-TBM, shield, flange, pipes and bio-shield plug for the nuclear analysis. These CAD data were converted to MCNP data by MCAM, and the MCNP data were inserted to the C-lite MCNP model. We conducted the shielding calculations in operation and after shutdown using this MCNP model, and evaluated the effective dose rate in the pipe forest area. We applied D1S MCNP for the calculation after shutdown. We adopted the following two cases. (1) We considered the particles passing through the gaps between the frame and the port wall (Case 1). (2) We killed the particles passing through the gaps between the frame and the port wall (Case 2).

The decay photon effective dose rates in 10^6 seconds after shutdown are 70 – 170 μSv/h in the pipe forest area in the Case 1, and these are 30 – 90 μSv/h in the Case 2. The decay photon effective dose rates after shutdown in the pipe forest area are lower than the target value of 100 μSv/h for hands-on access in the Case 2.

In addition, we evaluated the absorbed dose rate of silicon in electronics in the port cell due to photons from ^{16}N generated by the $^{16}O(n,p)$ reaction in the water. The absorbed dose rate is about 100Gy/h at 10 cm distance from the cooling water pipe. It is too high, and an additional local shield for electronics in the port cell needs to be considered.

X^th ITER Neutronics Meeting, Cadarache, France 30 June - 3 July 2015

Preliminary Calculations of Shutdown Dose Rate for the CTS Diagnostics System

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DTU and IST 2 are partners in the design of a collective Thomson Scattering (CTS) diagnostics for ITER through a contract with F4E. The CTS diagnostic utilizes probing radiation of ~60 GHz emitted into the plasma and, using a mirror, collects the scattered radiation by an array of receivers. Having a direct and unshielded view to the plasma, the first mirror will be subject to significant radiation and among the first tasks in the CTS design, is to determine whether the mirror will need active cooling.

At present the CTS is in the conceptual design phase and the related neutronics calculations focus on supplying input which affect the system design.

Examples include:

- Heatloads on plasma facing mirrors and preliminary stress and thermal analysis
- Port plug cooling requirements and it's dependence on system design (in particular blanket cut-out)
- Shutdown dose-rate calculations (relative analysis, depending on system design choices)

Latest Results from ICH Neutronics Analysis of New Port Gap Configurations

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The Ion Cyclotron Heating and Current Drive system (ICH) is designed to launch RF power into the ITER plasma, and will reside in equatorial port plugs (EPP) 13 and 15. Shutdown dose rates (SDDR) within the ICH port interspace are required to be less than 100 μSv/hr at 10^6 seconds cooling, in locations where maintenance access is required.

Streaming of neutrons down the gap between the port frame and the port extension significantly contributes to the SDDR in the port interspace, and previous neutronics analysis is now obsolete following the issue of PCR-439 which specifies increased gaps. Neutronics analysis was required for the updated ICH configuration in the C-lite reference MCNP model of ITER.

An accurate description of the blanket modules, manifolds and coils around the port plug is necessary to reliably model the neutron transport in the port gaps, however C-lite represents sector 1 of ITER, centred on an even port (above a lower port). To represent the local environment around the ICH port, the C-lite model was rotated to represent an odd port, and a novel technique used to update the in-vessel geometry locally around the EPP to correspond to the latest design.

Whilst designs are readily updated in CAD, modification of the corresponding MCNP geometry remains prohibitively effort-intensive and only simplistic modifications are realised within the scope of most neutronics analysis tasks. The approach taken here involves simplification and conversion of a subset of the geometry closest to the EPP - an updated representation of the local in-vessel environment at the level of the 'semi-detailed' models was then created in a fraction of the effort needed to model the entire in-vessel geometry.

Neutronics analysis for SDDR was then conducted for two configurations of front shims. This presentation will detail the modelling, calculation methodology and results.

Neutronics for ITER Diagnostic Upper and Equatorial Ports

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This talk presents an overview of neutronics analyses performed for Diagnostic Upper and Equatorial Port Plugs (UPP & EPP) of ITER, with their generic configurations and with particular diagnostic systems integrated in UPP#3, UPP#18, and EPP #17. The common neutronics problems and design solutions are traced along the talk. The ITER radiation requirements including human nuclear safety and machine reliability must be satisfied. This aim is achieved by developing the optimal shielding designs of the particular port plugs. For this optimization the objective function is the Shut-Down Dose Rate (SDDR) at Port Interspace (PI) and the target is minimization of SDDR. This presentation describes a number of design solutions for the port plugs (radiation stoppers, labyrinths, collimators, as well as selection of shielding and low activation materials). Due to possible universality of the ports, presented information might save R&D resources in future.

This talk is focused on the analyses of two diagnostics systems: 1) Charge eXchange Recombination Spectroscopy (CXRS) in UPP#3 – optical system with mirrors and bended channel, 2) Diagnostic Core-Imaging X-ray Spectrometer (CIXS) in EPP#17 – straight channels from plasma till crystals at PI. The CAD-based radiation transport computations have been performed with the Monte Carlo MCNP5 code and two methods of SDDR calculations: Direct 1-Step (D1S) and Rigorous 2-Step (R2Smesh). Using the outcomes of generic UPP & EPP analysis[3], the new neutronics results are presented in this talk, among them - SDDR at PI of EPP#17 with CIXS, that is reduced by 100 times: from 2 mSv/h in the original CIXS design to 20 microSv/h by means of additional B_4C shield blocks, reduction of the beam apertures, and beam collimators.

[3] A. Serikov et al., "Shut-Down Dose Rate analysis for ITER Diagnostic Equatorial and Upper Ports," *Fusion Engineering and Design*, **89**, pp. 1964–1968 (2014).

The Effect of Neutronics Model Simplification on Shutdown Dose Rate

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Promoted by ITER design, the methods for shutdown dose rate calculation, such as the rigorous two-step method and direct one-step method, have been greatly improved to approach higher calculation accuracy. However, the accuracy of calculation depends not only by the methods, but also by the precision of analysis model. In this study, taking the activation analysis for the first wall of No.15 blanket as an example, the effect of neutronics model simplification on shutdown dose rate calculation is analysed. Three neutronics blanket model with different simplification level are created by SuperMC/MCAM and integrated into the same ITER neutronics model, one is simplified plate model, one is detail model with material homogenization, and another model is precise in which the coolant channels and void gaps are descripted independently. The two-step method which couples the Monte Carlo transport program SuperMC with activation calculation program is used. The distribution of neutron spectrum, material activity, radio-nuclides, decay gamma intensity and spectrum is compared in detail.

It was shown that the model simplification, especially the major geometry simplification, has changed the distribution of neutrons. Comparing the three models, the difference between the detail model and precise model is small, except for locations behind water with increase of thermal neutrons. But for plate model, both the neutron intensity and decrease rate shows great difference.

Recent Developments in the Attila® Software of Significance to ITER Neutronics

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Recent developments in Attila will be presented, with an emphasis on features of significance to ITER neutronics. These developments are featured in Attila Version 9.0, which is slated for release this summer. The near term development roadmap will also be presented. Attila Version 9.0 supports a tight integration with MCNP 6.1. Through Attila, users can import CAD models and mesh them for the unstructured mesh capability in MCNP 6.1. Automated weight windows with consistent source biasing[1] is included for variance reduction, with global variance reduction[2] being added in Version 9.1 (autumn 2015 release). The Attila graphical user interface supports most aspects of the MCNP 6.1 calculation set-up, and duplication of effort between Attila and MCNP 6.1 is minimized. Attila will also split regions with many mesh cells into sub regions, keeping track of material, source and tally assignments, resulting in optimal MCNP6 unstructured mesh load-in times, tracking times, and memory requirements. Development and testing of the FENDL 3.0 data set is currently underway. In Version 9.0, building a cavity part to fill all empty internal regions is no longer required, as Attila will automatically mesh all enclosed volumes, and will assign each discrete volume a unique region name. Users can rename these regions, and assign material properties and sources definitions just like any other region. Automatic anisotropic curvature refinement has been implemented, which refines only in the curvature direction, and preserving a much larger axial element size. Improvements have also been added to the sweep ordering, including improved memory management and efficiency. Additional enhancements will also be presented.

1 J Wagner, A Haghighat, "Automated Variance Reduction of Monte Carlo Shielding Calculations Using the Discrete Ordinates Adjoint Function", Nuclear Science and Technology, Vol 128(2), Feb 1998.

2 D Peplow, T Evans, J Wagner, "Simultaneous Optimization of Tallies in Difficult Shielding Problems", Nuclear Technology, Vol 16 8(3), Dec 2009.

Attila Port Cell Analysis Using Previous Neutronics Results as Neutron Source Creation

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In ITER, both the interspace and port cell regions will have difficulties with neutronics. Especially when it comes to diagnostic components and the required feed through throughout the boundary locations. Using the Attila code, a highly resolved solution can be found within the interspace region. But due to the size of the required model, finding a highly resolved solution within the port cell takes excessive computational time. This problem is resolved by creating sub models of the port cell and interspace. The interspace model is used to solve for a detailed solution up to the port cell boundary, and that solution is then used for the port cell analysis. There is one difficulty that must be solved before a port cell sub model with the interspace neutron source can be used. Currently, the Attila neutronics code cannot handle boundary source creation from previous calculations. A method has been devised that will be used to create a viable and reliable boundary source for the port cell analysis. This boundary source is then used for detailed port cell analysis that incorporates changes to the bioshield plug and port cell components. This allows for quick accurate neutronics calculations that take into affect overall contributions all of ITER interspace region components. Once this method has been verified, then an iterative process will be created to speed up diagnostic implementation, which will include a detailed port cell plugs and diagnostic labyrinths.

A detailed solution has been obtained for the interspace region neutron flux contributions. This was done as a study between the two neutronics codes Attila and MNCP with the MCNP work done at the University of Wisconsin – Madison. This work has show that both codes give agreeable solutions for a large 40° ITER model that covers from the center of the machine to the bioshield. This report covers the initial results for the Attila boundary creation at the bioshield using the results from the Attila/MCNP analysis and the neutron flux contributions within a simplified port cell region.

This work is supported by DOE contract numbers DE-AC02-09CH11466 (PPPL) and DE-AC05-00OR22725 (UT-Battelle, LLC). The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Status of UW Code Developments and Physics Comparisons

A Davis, P PH Wilson, CNERG Team

Description will be given of the status of code developments at the University of Wisconsin-Madison, specifically addressing: updates to the DAGMC toolkit, the Once Through Irradiation System (OTIS) and recent PyNE enhancements. The main body of the presentation will focus on the comparison of a detailed calculation of DAG-MCNP5 & Attila done in collaboration with PPPL. The comparison will include details of: neutron fluxes, neutron heat, activation, 3D meshes and cell tally results are compared. The DAGMC comparison shows that when given similar approximations into each code good agreement can be seen between each code. Time will be dedicated to the results from the estimate of the N14/O17 concentration determines for the BM4 coolant loop with comparison made to the previous reference calculations.

iMesh – A Program for the Post Processing of VTK Radiation Maps

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The present work presents the development and capabilities of a computer program, named iMesh, for the post processing of radiation maps in vtk format such as those generated from MCNP simulations. The software has been developed within the ITER framework contract # 6000000141, task order #2[4] by the UNED-Idom Consortium.

Radiation maps in vtk format are read by the program and converted to a vtk unstructured grid by Delaunay tetrahedralization. Once converted, a set of operations can be performed on the input grids. These can be scaled, translated and rotated independently; their field can be normalised or modified using a flexible combination of polynomials and exponentials. Eventually, iMesh can perform mapping, replacement, sum or merge operations on the grids in order to obtain the resulting vtk file. One of the input vtk files (master) is taken as reference for these operations. More specifically, the mapping operation projects the field values on the master vtk grid, replacing the field values; the replace operation completely removes the cells of the master grid and substitutes them with those of the other grid; the sum operation is a mapping operation, but instead of replacing the field values, they are added; and finally, the merge operation is a special case of replace, where both grids are kept and the corresponding field values are interpolated between them.

The code, developed in C++, uses the open source vtk libraries[5] and has been compiled on Linux and Windows platforms. During development, standard methodologies and nuclear QA procedures have been used (including requirements gathering by use cases, source code management and version control, code peer-review and automatic regression testing).

iMesh represents a flexible and time saving tool for the combined post processing of different vtk radiation maps, thus obtaining more comprehensive data for radiation analyses.

[4] Technical Specification for Task Order 02 (Neutronics Framework Contract IO/13/6-141) - IDM UID PR29RS
[5] Release 6.1.0 of VTK was used. See www.vtk.org

Comparison of Codes for Simulation of Neutron Production in Accelerator Based DT Neutron Generators

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Accelerator based neutron sources will be used for the experimental calibration of neutron detectors in large tokamaks, so there is a need for computational tools capable of simulating their neutron production. These tools are necessary in order to understand the characteristics of such neutron sources and to provide reliable computational support for the calibration of JET's neutron monitors with 14 MeV neutrons that will be used as a benchmark for the neutron calibration of ITER.

A comparison of the results of the existing codes, capable of such simulation, was done in the preparation for the JET compact neutron generator characterisation experiments that will be carried out later in 2015. Accurate simulation of the generator is necessary both to plan these experiments and to support the analysis of their results. Advantages and disadvantages of the use of different codes (ENEA-JSI subroutine, MCUNED and DDT) will be discussed, results like angular dependency of neutron flux and spectra will be presented, and possible sources of the discrepancies explained. Future plans related to simulation and modelling of the neutron generator for its experimental characterisation will be described.

Development and Experimental Validation of Super Monte Carlo Simulation Program for Fusion Applications

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There are great challenges of Monte Carlo (MC) simulation codes for fusion applications, including the calculation modelling of complex geometries, simulation of deep penetration problem in radiation shielding, slow convergence of complex calculation, lack of experimental validation on nuclear performances of key fusion components, etc.

SuperMC is a general purpose, intelligent and multi-functional program for the design and safety analysis of nuclear systems. It is designed to perform the comprehensive neutronics calculation, taking the radiation transport as the core and including the depletion, radiation source term/dose/biohazard, material activation and transmutation, etc. It supports the multi-physics coupling calculation including thermo-hydraulics, structural mechanics, biology, chemistry, etc. The main technical features are hybrid MC-deterministic methods and the adoption of advanced information technologies. The main usability features are automatic modelling of geometry and physics, visualization and virtual simulation and cloud computing service. The latest version of SuperMC can accomplish the transport calculation of n, γ and can be applied for criticality and shielding design of reactors, medical physics analysis, etc.

For fusion applications, SuperMC has been validated by extensive benchmark models and experiments, including the International Thermonuclear Experimental Reactor (ITER) benchmark model, the experiments in the Shielding Integral Benchmark Archive Database (SINBAD). The verification results with the ITER benchmark model was introduced in this paper. The calculation results of SuperMC were consistent with the results of MCNP5. SuperMC can greatly reduce human efforts by directly using CAD models and the calculation efficiency is higher than that of MCNP5 by 55% on average. As the supplementary of validation experiments of MC software for advanced nuclear energy systems applications, experiment for deep penetration problem in radiation shielding and neutronics integral experiment of fusion blanket are being particularly conducted using High Intensity D-T Fusion Neutron Generator (HINEG) which will produce 14.1MeV neutrons.

Python Command Line Tool to Rename Cells, Surfaces, Universes and Materials in MCNP Input Files

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One of approaches to prepare a complex MCNP model is to develop input files for model parts, verify them separately and then integrate them into one model using the universe-and-fill mechanism. This approach is followed, for example, to develop the ‘C-Lite’ MCNP model of ITER, where semi-detailed representation of the component latest designs is developed by several organizations. Although this approach is conceptually straightforward, its implementation is usually tedious since all MCNP model elements must have unique numbers in the final model and it is not always possible to set up number ranges to be applied for each model's part in advance. Generic plain text editors are not equipped with tools to tackle this problem, so that renaming of cells, surfaces and materials becomes one of the most time-consuming operations. To overcome this problem, we have developed a tool with the following features:

- The input file structure, including comments and line wrapping is preserved, so that introduced changes can be controlled by common tools, e.g. ``diff`` or ``vimdiff``.
- Different mapping rules can be applied to different ranges of numbers. Thus, one can change only a subset of numbers, leaving all other unchanged.
- If necessary, lines in the renumbered input file can be wrapped to obey the 80 characters limit.
- A log file containing all substitutions can be used as an input to perform reverse change. This is useful to verify results: the input file obtained by the reverse renumbering should be identical to the original input file.
- The tool is written in Python and is cross-platform. Its command line interface can be run on a remote machine with terminal access and does not require Pyhton knowledge.

This presentation describes current possibilities of this tool and proposes possible usage scenario for handling large MCNP models like ‘C-Lite’.

The CAD to MC Conversion Tool McCad: Recent Advancements and Applications

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McCad is a geometry conversion tool developed at KIT to enable the conversion of CAD and Monte Carlo (MC) geometry models for neutronics calculations (CAD to MC). The kernel of conversion algorithm includes the convex solids decomposition and void spaces filling [1].

This work is devoted to an advanced version of McCad based on the implementation of an improved decomposition algorithm, in order to improve the efficiency and reliability of the conversion process, and avoid the programme errors arise from the frequent Boolean operations employed by the original decomposition algorithm. First, the collision detection method based on triangles is employed for classifying the splitting and non-splitting boundary surfaces of input solids, and then sorting algorithms of splitting surfaces by means of features recognition technology are introduced for calculating the sequence of splitting surfaces. Finally, the input solid with complex geometry is decomposed into a combination of simple solids with the sorted splitting surfaces. Compared with original one, the new decomposition result is simpler and more regular. In the meantime, upgraded graphic kernel OCC (OpenCascade) [2] is employed by new version of McCad for decreasing the potential Boolean operation errors, and a brand new architecture of software is also constructed for advancing the openness and Scalability.

In addition, a new graphic interfaces based on Salome open source platform is being developed and a preliminary version has been released, which supplies more user friendly interface for visualization, geometry modeling, interactive operations etc. And it makes the pre-process of geometry and conversion of models more flexible and convenient.

The new version of McCad with new algorithms and features are being tested and have been successfully applied for the CAD to MC conversion works on upgrade NBI port [3], blanket for ITER and new DEMO design, performed in the frame of the EFDA PPPT (Power Plant Physics and Technology) programme [4].

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Advanced MC Modeling and Multi-physics Coupling System

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An advanced system has been developed at the Karlsruhe Institute of Technology (KIT) for complex Monte Carlo (MC) geometry modelling and multi-physics coupling. In this system, an MC modelling module provides the conversion of CAD geometries to hybrid Constructive Solid Geometry (CSG), tessellated solids and mesh MC geometry representations. A generic multi-physics coupling module provides data mapping and interfacing for the MC codes MCNP5/6, TRIPOLI-4, Geant4, the CFD codes Fluent and CFX, and the Finite Element (FE) simulation platform ANSYS Workbench. These two modules have been integrated into the open-source simulation platform SALOME which provides them with CAD modelling, mesh generations and data visualizations capabilities.

This advanced system has been verified by a series of test cases, e.g. ITER Benchmark, Helium Cooled Pebble Bed (HCPB) Test Blanket Module (TBM), etc. The reliability and efficiency of the system were concluded.

Generating of Fusion Plasma Neutron Source with AFSI for Serpent MC Neutronics Computing

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Generating of a realistic neutron source for introducing the Serpent MC code [1] in fusion applications is performed with the AFSI Fusion Source Integrator [2]. AFSI is working as a part of the ASCOT Monte Carlo fast particle following code package [3, 4] calculating tokamak fusion reactivity and distribution of the reaction products, and it is suitable for defining neutron production in all birth channels (thermal-thermal, thermal-fast and fast-fast particle reactions). An ITER baseline Q=10 plasma with D/T mix (50%/50%) will be presented as a demonstration case, as, where the neutron production rate is given by AFSI in (R, z) coordinates.

Benefits of AFSI compared to previously applied methods based on simplified analytical approximations of plasma parameters, such as T and n, include better accuracy of the source geometry and possibility to include all reaction types to the analysis. In addition, AFSI is capable of coupling the neutron source definition to time-dependent plasma transport simulations by, for example, the ETS [5] or JINTRAC [6] packages, which is useful in analysis of yet non-existing devices, such as ITER and DEMO.

Several development steps in the AFSI-based neutron source have been planned. In the current version of the code, only thermal particle reactions have been taken into account, and flux surface symmetry is used there. Generating anisotropic (thermal-fast and fast-fast particle reactions) fusion reaction product distributions (location coordinates and energy included) is under construction. Additionally, neutron source is defined with (R,z) coordinates in AFSI and this will be updated to enable selection between (ρ,ψ) and (R, z) coordinates, which makes testing of the efficiency of the method and its sensitivity to small variations in source geometry or plasma parameters more fluent.

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[3] J. A. Heikkinen et al. 2001 Journal of Computational Physics 173 527-548.
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[5] D. P. Coster et al. 2010. E IEEE Transactions on plasma science 38 9.
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On the Use of the Serpent 2 Monte Carlo Code for Fusion Applications

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The Serpent Monte Carlo code has been developed at VTT Technical Research Centre of Finland since 2004, with a specific focus on fission reactor physics applications. The code reads continuous-energy interaction data from ACE format cross section libraries and features a built-in depletion solver for simulating the isotopic changes in materials subject to neutron irradiation. Serpent has an international user community consisting of more than 400 users in 131 universities and research organizations in 35 countries around the world. The traditional applications range from fuel assembly-level reactor physics and burnup calculations to research reactor modelling and coupled multi-physics applications.

In recent years, some considerable effort has been devoted to coupled calculations and the development of a universal multi-physics interface designed to exchange data with fuel performance, system-scale thermal hydraulics and CFD codes. The CFD code coupling is based on the OpenFOAM file format and an unstructured polyhedral mesh. The work on multi-physics coupling required implementing entirely new geometry routines, which consequently lead to the development of new geometry types, capable of handling complicated irregular structures that are difficult to model using the conventional CSG geometry type. Including gamma heating in coupled calculations requires development of a photon transport mode, which is currently under way.

Even though Serpent is traditionally characterized as a reactor physics neutron transport code, the recent developments in methodology have raised the interest in broadening the scope of applications beyond fission reactor analysis, in particular radiation shielding and fusion neutronics. In practice, Serpent could be used for heat deposition, material damage, activation, tritium breeding and shut-down dose rate calculations. The features and capabilities relevant for fusion neutronics applications are introduced and discussed. The advanced CAD-based geometry capabilities in Serpent 2 are demonstrated with an example involving the C-Lite model of the ITER reactor.[6]

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6 J. Leppänen, "*CAD-based Geometry Type in Serpent 2 -- Application in Fusion Neutronics.*" In proc. *M&C + SNA + MC 2015* Nashville, TN, Apr. 19-23, 2015.

Development of D1S-UNED Package for Shutdown Dose Rate Calculations

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The TECF3IR neutronics group at UNED has developed the D1S-UNED package based on D1S methodology, consisting in a modified version of MCNP5 (MCNP-D1S) plus two scripts devoted to produce data necessary for the simulation.

Code systems based on D1S methodology for shutdown dose rate calculation are very useful tools for neutronics studies in the design of components to be implemented in ITER. Two main advantages of D1S over R2S systems are: i) Significantly shorter times and computational resources consumption, and ii) absolute spatial and energy resolution of the neutron flux. In return, the setup of the calculation is more cumbersome since it requires identification of the most contributing isotopes, use or production of specific transport libraries, and, calculation of a time correction factor for each contributing isotope which is related to the neutron irradiation scenario.

Relevant D1S-UNED features are: i) 3D maps production of shutdown dose rates and decay gamma sources, ii) filtering options to identify specific contributions from isotopes and/or cells, iii) generation of modified D1S-transport libraries using data from activation files, and iv) possibility to define different materials definition in cells depending whether neutron or photon are transported

In this package, a python script is devoted to generate D1S libraries from transport library in ENDF format, in which gamma production cross sections are substituted by decay gamma production cross section. These gamma production cross sections can be generated using activation cross sections defined in original transport library or using external cross section data. Any transport and activation libraries (in ENDF and EAF-pointwise format respectively) can be used to generate D1S libraries. The current set of D1S nuclear data has been generated with FENDL2.1 and EAF2007.

Finally, a second python script has been also developed to provide the isotope time correction factors to be used in D1S methodology depending on the irradiation scenario. For a given irradiation scenario, the script solves analytically the evolution equation of the daughter isotope considering only the first direct parent-daughter activation reaction.

Since the use of such methodology can be laborious for unexperienced users, the package has been developed such way an intermediate MCNP user can easily manage the system.

Combined Neutron Gamma Analysis with CONGA: Status of Development

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Neutron activation is a key process that needs to be considered for various ITER neutronics analyses. It is required, for instance, to compute the shutdown dose rates where the transport of photons from activated components needs to be calculated. A well-established method of simulating the combined neutron and photon physics associated with neutron activation is the Rigorous 2-Step Method (R2S method): In the first step, neutron flux spectra are calculated within a mesh on the geometry under consideration. The spectra are then used as the basis for activation calculations resulting in photon spectra throughout the geometry. The second step involves the transport of the secondary photons and the calculation of various mesh tallies.

Amec Foster Wheeler have developed the neutronics analysis software CONGA which implements the R2S method. Currently, MCNP and FISPACT are used for the neutron/photon tracking and the activation calculation, respectively. The computation is based on a system of two spatial meshes overlaid on the geometry. The neutron and photon spectra are evaluated on the coarse mesh, whilst a fine mesh offers the possibility to optimise the calculation by taking into account a detailed spatial distribution of the neutron flux. CONGA has a flexible, script-driven architecture, with minimised modification to the MCNP code and MPI based parallelisation of the FISPACT calculations, enabling future extensions of the software capabilities.

In order to validate CONGA, various test and benchmark cases have been run including an experiment involving a stainless steel/water assembly carried out at the 14MeV Frascati Neutron Generator and dedicated to the validation of dose rate calculations for ITER. This presentation involves an overview of the structure of CONGA and the status of its development as well as the discussion of the validation cases considered.

Comparison of SuperMC and MCNP Calculation Capabilities Using the ITER Neutronics Model

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SuperMC is a general purpose, intelligent and multi-functional program for the design and safety analysis of nuclear systems. The latest version of SuperMC can accomplish the transport calculation of n, γ and can be applied for criticality and shielding design of reactors, medical physics analysis, etc.

The ITER neutronics model, created by ITER International Organization, was used for testing and comparison of CAD/Monte Carlo codes developed by different institutes and universities. 8 calculation cases of results have been calculated and analysed: (1) neutron wall loading on the first wall, (2) neutron flux on the divertor surface, (3) neutron flux in the divertor cassette, (4) nuclear heat in the divertor cassette, (5) nuclear heat in TF coils, (6) neutron flux in dummy port plug, (7) neutron flux in shield plug and (8) neutron flux in tally spheres. These 8 cases of calculation results of SuperMC were well consistent with the results of MCNP within the statistical uncertainty inherent in the MC method. SuperMC is faster than MCNP by 55% on average for these 8 calculation cases.

In testing with ITER neutronics model, the major advantages between SuperMC and MCNP are following: (1) Generally, SuperMC directly starts from importing CAD models and converts geometry internally for further particles transport calculation while MCNP calculation starts from the ASCII text input file. (2) Complex spatial distribution of sources such as plasma source in ITER neutronics model can be converted from CAD model and probability distribution can be assigned in visualized and interactive manner. (3) SuperMC adopts hierarchical solid geometry description method assisted with surface description method while MCNP mainly adopts surface description method. (4) With some acceleration methods, SuperMC is faster than MCNP for these 8 calculation cases of ITER neutronics model.

Advanced Algorithms of SuperMC/MCAM and Its Application in ITER Clite Model

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SuperMC/MCAM, the Multiple Physics Coupling Analysis Modelling Program was designed to realize the CAD based automatic MC modelling. The latest released version SuperMC/MCAM5.2 supports bi-directional conversion between CAD models and multiple Monte Carlo code, such as SuperMC, MCNP, FLUKA, Geant4 and Tripoli. It has been updated to the latest ACIS platform and supports x64 as target operation system which greatly increased the supported model scale. Based on this version, two new methods were developed to further support more detailed and complex models such as ITER Clite model.

For MC codes, all the space including solids and void spaces between them need to be described considering accuracy and efficiency. A dedicated void filling method named Convex-based Void Filling (CVF) was proposed in this paper. This method subdivides all the transport space into sub-regions iteratively with high quality subdivisions and generate the description of void spaces by complementary describing the volumes in the sub-regions.

As the system of abundant parts were usually modelled in modules, the traditional way that treat all parts in flat structure is not suitable for hierarchically designed complex systems anymore. A new multiple level hierarchical structure for parts description in MC geometry modelling has been implemented. This structure enables convenient replacement of geometric objects in specified region in CAD model and the converted MC model which would improve the modelling efficiency.

The new methods have been implemented in SuperMC/MCAM5.2 and tested with neutron transport calculation of International Thermonuclear Experimental Reactor (ITER) Clite model. Results demonstrated high efficiency of the proposed method for both geometry converting and MC calculation.

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Benchmark Experiment on Molybdenum with DT Neutrons at JAEA/FNS

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Type 316 stainless steel (SS316) includes a few percent of molybdenum in order to improve corrosion resistance. A benchmark experiment on molybdenum is performed with DT neutrons at JAEA/FNS in order to validate recent nuclear data of Mo. A rectangular Mo assembly is placed at a distance of 201 mm from the DT neutron source. The size of the Mo assembly is 253 mm × 253 mm × 354 mm. The Mo assembly is covered with 51, 202 and 253 mm thick Li_2O blocks around the front, side and back surfaces, respectively. The Li_2O blocks eliminate background neutrons at the measuring points in the Mo assembly. Several dosimetry reaction rates and fission rates are measured, and the measured values are compared to calculated ones with a Monte Carlo transport code MCNP5-1.40 and recent nuclear data libraries of ENDF/B-VII.1, JEFF-3.2 and JENDL-4.0 (FENDL-3.0). The ratios of the calculated values to the experimental ones (C/Es) generally decrease with the increasing distance from the front surface of the Mo assembly. It is considered that the molybdenum data in the recent nuclear data libraries have some problems.

Validation of IRDFF with Graphite Integral Experiment at JAEA/FNS

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Recently the International Atomic Energy Agency (IAEA) has released International Reactor Dosimetry and Fusion File release 1.0 (IRDFF 1.0), which is a successor of the International Reactor Dosimetry File (IRDF-2002), in order to cover fusion applications in addition to reactor dosimetry. The Nuclear Data Section (NDS) of IAEA has initiated a Co-ordinated Research Project (CRP) in order to validate the IRDFF 1.0 library. Under the CRP, we perform an integral experiment with a pseudo-cylindrical graphite assembly and DT neutrons at the Fusion Neutronics Source facility of Japan Atomic Energy Agency (JAEA/FNS). Note that the neutrons inside the graphite assembly are not only DT neutrons but also neutrons below 14 MeV including a Maxwell peak. We can validate the IRDFF data even for the capture reactions with this experiment. Forty dosimetry reaction rates are deduced by measuring decay gamma-rays from a lot of foils placed in the assembly with HPGe detectors after DT neutron irradiation. The measured reaction rates are compared with calculated ones with a Monte Carlo transport code MCNP5-1.40, a nuclear data library ENDF/B-VII.1 and IRDFF-v.1.03. The calculated results are consistent with the experimental ones. It supports that IRDFF has no big problems.

Analyses of Iron and Concrete Shielding Experiments at JAEA/TIARA with JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0

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IAEA released a new Fusion Evaluated Nuclear Data Library, FENDL-3.0, in December of 2012. FENDL-3.0 extends the neutron energy range from 20 MeV to 150 MeV. Now there is increasing interest in nuclear data above 20 MeV. Thus we have analyzed the iron and concrete shielding experiments with the 40 and 65 MeV neutron sources at TIARA in Japan Atomic Energy Agency with the latest high-energy nuclear data libraries, JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0.

The Monte Carlo code MCNP-5 and ACE file of JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0, which are supplied from JAEA, BNL and IAEA, respectively, were used for this analysis. The collimated neutron beam and experimental assemblies were modelled in the analysis. The measured source neutron data were adopted in the analysis. The followings are found out from the results; 1) the calculations with JENDL/HE-2007 agree with all the measured ones well, 2) those with ENDF/B-VII.1 tend to overestimate the measured ones with the thickness of the assemblies largely, 3) those with FENDL-3.0 agree with the measured ones well for the iron experiment, while they overestimate the measured ones well for the concrete experiment largely. Some data in ENDF/B-VII.1 and FENDL-3.0 should be revised.

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Status of KERMA and DPA in FENDL-2.1

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The Fusion Evaluated Nuclear Data Library version 2.1 (FENDL-2.1) is used as the standard nuclear data library for ITER neutronics analyses. We can perform correct radiation transport calculations with FENDL-2.1. However little is known about problems of the KERMA factors and DPA cross section data in ACE and MATXS files of several nuclei in FENDL-2.1. These are due to incomplete nuclear data. Particularly the KERMA factors are not correct at all for nuclei with inconsistent energy balance. In this presentation we will show the status of the KERMA factors and DPA cross section data in ACE and MATXS files of FENDL-2.1 and would like to invite your attention to this issue.

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Problems on FENDL-3.0

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A new version of Fusion Evaluated Nuclear Data Library (FENDL), FENDL-3.0 was released from IAEA in 2012. FENDL-3.0 has the following features; 1) extension of the neutron energy range of neutron-induced reactions from 20 MeV to more than 60 MeV, 2) activation data libraries for proton- and deuteron-induced reactions up to more than 60 MeV. We carried out the benchmark tests of the general-purpose data library for neutron-induced reactions in FENDL-3.0 with the integral experiments at JAEA/FNS, JAEA/TIARA and OKTAVIAN. We also tested the MATXS files of FENDL-3.0 with a simple calculation model and compared KERMA and DPA data included in the ACE and MATXS files of FENDL-3.0 with those in other nuclear data libraries. In this symposium we present the following problems in FENDL-3.0 found out in our study; 1)^{16}O data above 20 MeV, 2) MATXS files above 20 MeV, 3) KERMA and DPA data included in the ACE and MATXS files. These problems were reported to IAEA last year. IAEA will improve these problems.

Overview of FNS Activities - Tritium Recovery Experiment -

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We have performed various neutronics researches; neutronics experiments with the Fusion Neutronics Source (FNS) facility in Japan Atomic Energy Agency, ITER nuclear analyses and other related issues. Since the last ITER neutronics meeting, the followings have been carried out;

1) Nuclear analyses for Japan Test Blanket Module
2) Tritium recovery experiment
3) Integral experiment for molybdenum nuclear data benchmarking
4) Graphite experiment for IRDFF benchmarking
5) Analyses of JAEA/TIARA shielding experiments
6) FENDL-2.1 KERMA and DPA status
7) FENDL-3.0 problems

We present the overview of these topics and our research situation which will be drastically changed in April, 2016.

Radiation Damage and Nuclear Heating Studies for the JET DT Campaign

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A new Deuterium-Tritium campaign (DTE2) is planned at JET in 2017, with a proposed 14 MeV neutron budget of $1.7\cdot10^{21}$, which is nearly an order of magnitude higher than in previous DT campaigns. It is estimated that a total neutron fluence on the first wall of JET of up to 10^{20} n/m^2 could be achieved, which is comparable to the fluence occurring in ITER at the end of life in the rear part of the port plug, where several diagnostic components are located.

The purpose of the present work is to evaluate the material damage and nuclear heating of selected structural materials to be tested at JET during DT plasma operation. These quantities are calculated with two methods; with the use of the MCNP code and with a combination of MCNP and FISPACT, and compared. A comparison of the appropriate nuclear data libraries is also presented.

The emphasis is on the long term irradiation stations, located close to the first wall at outboard midplane, offering the opportunity to irradiate samples of functional materials used in ITER diagnostics, to assess the degradation of the physical properties. The radiation damage and the nuclear heating were calculated for selected materials irradiated in these positions and for the neutron flux and fluence expected in DTE2. The studied candidate functional materials include, among others, Sapphire, YAG, ZnS, Spinel, Diamond. A damage in excess of 10^{-5} DPA is estimated for the materials at in-vessel irradiation locations for the duration of the DT campaign.

Selection of SINBAD Benchmark Experiments for ITER Nuclear Analysis Validation

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ITER as a nuclear installation in France is governed by the INB Order of 7. February 2012, which demands amongst others that for the nuclear safety demonstration verified and validated calculational and modelling tools are used. Recent more focused interest on radiological protection now highlighted the need to define a framework in which ITER nuclear analysis can be validated compliant with the INB Order.

The Shielding Integral Benchmark Archive Database (SINBAD) is a collection of around 100 high quality experiments for validating and benchmarking computer codes and data for radiation transport and shielding applications. A coordinated effort at ITER and F4E, supported strongly by the SINBAD Fusion Task Force, was undertaken to identify key parameters, relevant experiments and elements of a validation procedure for ITER nuclear analysis. The paper will present a proposed set of fusion relevant SINBAD experiments for testing neutron/gamma transport codes. The outline of a general procedure for validation of transport, activation and coupled transport-activation problems will be described, which will encompass other experimental as well as computational benchmark.

Detailed 3-D Nuclear Analysis of ITER Blanket Modules and the Impact of Updated Cross Section Libraries on Neutronics Calculations

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The work presented in this talk will consist of two parts. Part 1 will present 3-D CAD based blanket and VV nuclear analysis work. Part 2 will present the impact of updated cross section libraries on ITER neutronics calculations.

Part 1: 3-D CAD based blanket and VV nuclear analysis

In this work we used the CAD based DAG-MCNP5v160 transport code to analyze detailed models inserted into a 40 degree partially homogenized ITER global model. The regions analyzed include BM08, BM14NDL near the heating neutral beam injection (HNB) port, and a 40 degree NBI sector with complete rows of detailed BM13-BM16 CAD models inserted. Nuclear heating in the vacuum vessel (VV), shield blocks, first wall (FW) fingers and beam were determined. DPA in the VV is also determined in some cases. Nuclear heating was mapped to ANSYS finite element meshes for subsequent thermal analysis.

Part 2: Impact of Updated Cross Section Libraries on ITER Neutronics Calculations

The accuracy of Monte Carlo transport codes such as MCNP is dependent on having accurate neutron and photon cross section data. Recently, an improved photon cross section library (eprdata12) has been made available with the MCNP6 Monte Carlo transport code. Additionally, an improved neutron cross section for fusion applications (FENDL3) has been undergoing benchmark testing for fusion applications.

In this work, neutronics results using the updated neutron and photon cross section libraries will be presented. This includes neutron and photon fluxes and heating calculations for a detailed 1-D cylindrical benchmark model of ITER. Additionally, nuclear heating results for actual components in a 40 degree 3-D detailed CAD model of ITER will be presented.

Preliminary Results of the Copper Benchmark Experiment at FNG

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A neutronics benchmark experiment[7] on a pure Copper block (dimensions 60x70x70 cm^3), aimed at testing and validating the recent nuclear data libraries for fusion applications under 14 MeV neutrons irradiation, was performed in the frame of a F4E specific grant at the 14-MeV ENEA Frascati Neutron Generator (FNG).

The relevant neutronics quantities (e.g. reaction rates, neutron flux spectra, doses, etc.) were measured using different experimental techniques to get the nuclear quantities of interest and the results were compared with the calculated quantities using fusion relevant nuclear data libraries.

This work reports the preliminary results of the post analyses carried-out by ENEA for activation foils measurements. Several activation foils (Nb, Al, Ni, In, Au and W) were placed at eight different radial positions along the block and irradiated with 14 MeV neutrons. The simulation of the experiment was carried-out using MCNP5 Monte Carlo code using JEFF 3.1.1, JEFF 3.2 and FENDL 3 nuclear data for neutron transport and IRDFF v1.05 library for the reaction rates in activation foils. A detailed 3D MCNP model of the Copper block and experimental assembly was integrated in the reference FNG MCNP model. The calculated (C) reaction rates in activation foils and neutron spectra were compared with the experimental quantities (E) and the C/E ratio with relative uncertainties was assessed. Calculations show an overall underestimation of the measurements, within 15% for high threshold reactions and more severe low threshold (E<0.1 MeV) and thermal reactions (within a factor 2-3).

[7] This activity was carried-out within F4E FPA-395

Recent Developments in the TRIPOLI-4 Code for Fusion Applications

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TRIPOLI-4® is a three-dimensional and continuous energy Monte Carlo particle transport code, developed by CEA, and dedicated to shielding, reactor physics with depletion, criticality safety and nuclear instrumentation. TRIPOLI-4® is currently able to simulate four kinds of particles:

- Neutrons from 20 MeV down to 10-5 eV,
- Photons from 50 MeV down to 1 keV,
- Electrons and positrons from 100 MeV down to 1 keV.

The TRIPOLI-4 version 9 (released in 2013) [1] already includes some features especially conceived for the fusion community, such as the possibility to define particle sources by means of subroutines (without recompiling the code) and the addition of the torus volume in TRIPOLI-4® geometry. Another important point to notice is the existence of external tools to convert CAD model into TRIPOLI-4® geometry: MCAM developed by FDS Team and McCad developed by KIT.

For the TRIPOLI-4® version 10 (release planned for December 2015), a new option has been added concerning the energy interpolation mode used for the neutron exit energy distribution. This option offers the users the opportunity to choose between the raw evaluation and a more physical treatment of these data (unit-base interpolation). Thus, it is now possible to quantify the impact of such an interpretation on high energy neutron and coupled neutron-gamma sources for fusion analyses (the most sensitive configurations)

Moreover, an activation scheme is being developed in TRIPOLI-4® to calculate shutdown dose rates for fusion applications. It is based on the Rigorous-Two-Steps (R2S) based on the Monte-Carlo code TRIPOLI-4® coupled with the depletion code MENDEL.

1. TRIPOLI-4® Monte-Carlo code performs a neutron transport calculation in order to compute the flux in each region susceptible to produce decay gammas. MENDEL depletion code computes the nuclide inventories and the decay photon sources for each region based on the neutron fluxes previously calculated by TRIPOLI-4®.
2. TRIPOLI-4® transports the decay photons and computes the dose rates induced in each region of interest.

Reference:

[1]: http://www.oecd-nea.org/tools/abstract/detail/nea-1878/

Proposal for a Complete Management Framework for ITER Nuclear Analyses

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ITER as a nuclear installation is governed by the French *INB Order of 7 February 2012*, which demands verified and validated calculations and modelling tools for safety demonstration. ITER nuclear analyses are highly complex activities, in many cases considered as protection important and/or integration, affecting very many systems at once. Accurate, detailed and extensive geometry models are required to obtain reliable answers. Moreover, conventional modelling and analysis tools used in the industry lack the necessary physics algorithms and/or nuclear data to tackle certain fusion problems. Finally, the conduction of ITER nuclear analyses is currently scattered between IO-CT teams, IO-DA teams, and contractors of both.

In view of such challenging environment, to satisfy INB requirements, and to ensure the reliability and traceability of such large amount of complex and de-centralised analyses, it is necessary to establish a robust and comprehensive framework for their management throughout the project. This framework shall include a number of validation and verification aspects like code benchmarking and independent review of models and calculations. Currently available reference models, standards and recommendations for neutronics analyses provide a significant head start, as they are already widely known and used by analysts since early in the project.

We present a proposal for completing such framework based on three sets of documents:

- Prescriptions for the validation of tools. This includes a largely experimental test suite for neutron, prompt photon, material activation and decay photon transport capabilities of codes in the energy and material ranges of relevance to ITER.
- Guidelines for the management of neutronics analysis including their preparation, conduction, reporting, verification and archiving.
- Suite of reference inputs and standards. This includes validated fusion-relevant nuclear data libraries, verified geometry models (e.g. buildings, tokamak components, others), sources (plasma, activated water, activated components, others), material definitions, operation scheme, response functions, safety factors and others. This suite shall be put under configuration control.

The framework must be supported by a digital platform in which neutronics management documents, models and analyses are stored for review, traceability and dissemination. The already available ITER engineering database, currently used for reference models, can be adapted for these purposes and its use prescribed throughout the project.

It is crucial for the success of ITER that this complete framework is finalised, reviewed, approved, disseminated and consistently used throughout the ITER project (IO-CT, IO-DAs, contractors) as a matter of urgency. We invite the ITER neutronics community to critically review and discuss this proposal.

Safety Factor Determination

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The French INB Order requires that the nuclear safety demonstration is based on "*appropriate, explicit and proven methods, which integrate assumptions and rules adapted to the uncertainties and to the sphere of knowledge of the phenomena involved*". One aspect of this is explicit determination of the safety factors which are recommended to be used with radiation transport calculations.

The philosophy of the use of safety factors will be discussed, a definition of the safety factor will be given and application to ITER will be given.

This presentation will provide a justification of the methodology used to determine safety factors using the gamma dose rate from the tokamak cooling water systems will be as an example.